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Formerly Utilized MED/AEC Sites Remedial Action Program

Radiological Survey of the Building Site 421, United States Watertown Arsenel, Watertown, MA

February 1980

Final Report

Prepared for

U.S. Department of Energy
Assistant Secretary for Environment
Division of Environmental Control Technology

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Washington, D.C. 20545

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PREFACE AND EXECUTIVE SUMMARY

This is one of a series of reports resulting from a program initiated in 1974 by the Atomic Energy Commission (AEC) for determination of the condition of sites formerly utilized by the Manhattan Engineer District (MED) and the AEC for work involving the handling of radioactive materials. Since the early 1940's the control of over 100 sites that were no longer required for nuclear programs has been returned to private industry or the public for unrestricted use. A search of MED and AEC records indicated that, for some of these sites, documentation was insufficient to determine whether or not the decontamination work done at the time nuclear activities ceased is adequate by current guidelines.

This report contains the results of surveys of the current radiological condition of the Building Site 421, United States Arsenal Watertown, Watertown, Massachusetts. Findings of this survey indicate there are four spots involving an area of less than 6 000 cm^2 of identifiable low-level residual radioactivity on the concrete pad which is all that remains of Building Site 421. The largest spot is approximately 5 000 cm^2. The other three spots are 100 cm^2 or less. The beta-gamma readings at these spots are 8.4×10^2 dis/min-100 cm^2, 2.2×10^5 dis/min-100 cm^2, 2.2×10^5 dis/min-100 cm^2 and 8.5×10^4 dis/min-100 cm^2. No alpha contamination was found at these locations. Gamma spectral analysis of a chip of contaminated concrete from one of the spots indicates that the contaminant is natural uranium. This contamination is "fixed" in the concrete and does not present an internal or external exposure hazard under present conditions. A hypothetical hazard analysis under a conservative set of assumed conditions indicates minimal internal hazard.

The highest End Window contact reading was 0.09 mR/h. None of the other three spots indicated an elevated direct reading with the End Window Detector.

Radon daughter concentrations were determined at three locations on the Building 421 pad. These were 0.00013 WL, 0.00011 WL and 0.00009 WL. According to the Surgeon General's Guidelines found in 10 CFR 712, radon daughter concentrations below 0.03 WL do not require remedial action in structures other than private dwellings and schools.

Soil samples taken about the site indicate no elevated levels above the natural background levels in the soil. A gamma spectral analysis of a water sample obtained from the storm sewer line near the Building 421 pad indicates no elevated radioactivity in the sample. It was therefore felt that no contamination is present in this sewer.

This radiological assessment was performed by the following Health Physics personnel of the Occupational Health and Safety Division, Argonne National Laboratory, Argonne, Illinois: R. A. Wynveen, W. H. Smith, C. Boggs Mayes, P. C. Gray and D. W. Reilly.

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INTRODUCTION

During the Manhattan Engineer District/Atomic Energy Commission (MED/AEC) Era, an area known as Building 421 was occupied on the United States Arsenal Watertown Site. The site was used a laboratory and office areas for the preparation and processing of uranium. Originally, laboratory test work was conducted at Massachusetts Institute of Technology (MIT) and in 1946 the laboratory was moved to the Watertown Site. Since there are no records of a final survey of the Building Site 421, it was decided to undertake a survey of the remaining floor pad of this building. All that is left of the Building 421 is a large concrete pad which had been the foundation of the building. The area of the pad was approximately 22 630 m^2. At the time of the survey, a portion of the pad was used as a storage area for concrete casket vaults. The area covered by these vaults was approximately 5% of the total area available for survey. The arsenal property east of the Main Gate, which includes the Building 421 site, is at present owned by the Watertown Redevelopment Authority, Watertown, Massachusetts.

SURVEY AND ANALYTICAL TECHNIQUES

General

All accessible concrete pad areas were surveyed, with the exception of that area which was utilized for storage of the concrete vaults. In addition, the exterior south wall of Building 331, since it was adjacent to Building 421, was surveyed its entire length to the height of 2 meters. See Table 1 and Figures 1A and 1B for locations of the areas surveyed. (All measurements were originally taken using the English system. The Standard International (SI) units are to the nearest approximation. For example, 2 inches are equal to 5 cm.)

Instrumentation

Four types of survey instruments were used (Table 2). An Eberline floor monitor, Model FM-4G, having a detection area of 325 cm^2, utilizing the Eberline PAC-4G-3 electronics, was used to survey the pad. A PAC-4G-3 with a hand-held detector, 51 cm^2 sensitive area, was used to survey the walls and other areas not accessible with the floor monitor. Double aluminized mylar (0.85 mg/cm^2) windows were used in both detectors. This allows for both low energy beta detection and increased instrument sensitivity. Both of these instruments were initially used in the beta mode. In the beta mode, the detector responds to alpha and beta particles and X and gamma rays. When areas were found which indicated a higher count rate than the instrument background, the beta reading was recorded and the instrument was then switched to the alpha mode and a reading of the alpha activity was obtained. In the alpha mode, the instrument only responds to particles with high specific ionization, such as alpha particles.

After the beta and alpha readings were obtained, an Eberline 530 End Window Geiger-Mueller (GM) Detector, with an Eberline HP-190 probe was used to obtain a reading of the contaminated area at contact. This instrument was also held one meter above the floor and was used to determine general ambient background radiation levels throughout the surveyed area. In addition, a Health Physics Instrument Model HPI-1010, which is a low range integrating ion chamber, was used at selected locations about the pad to determine background readings. This instrument was used in the integrating mode in order to obtain an integrated reading from several locations.

The HPI-1010 was calibrated using the gamma emissions from a Cobalt-60 (^{60}Co) calibration source. The End Window GM detector was calibrated using the gamma emissions from a Radium-226 (^{226}Ra) calibration source. The PAC-4G-3 instruments were calibrated in the alpha mode using a flat plate, infinitely thin Plutonium-239 (^{239}Pu) standard and in the beta mode with a flat plate, infinitely thin Strontium-90-Yttrium-90 (^{90}Sr-^{90}Y) standard. The PAC-4G-3 instruments were calibrated so as to provide an apparent 50% geometry.

It should be realized that the numerous isotopes which could be encountered will exhibit emission energies differing from that of ^{239}Pu and ^{90}Sr-^{90}Y utilized in the calibration. When detecting known isotopes that emit alpha and beta energies differing from that of the standards, a conversion factor for the particular isotope is developed to determine the appropriate yield.

When possible, the isotopes of contamination were identified. This was done by performing a gamma spectral analysis on a contaminated item or a representative sample from a contaminated area. A Nuclear Data Multichannel

Analyzer Model ND-100, utilizing a 10.1 cm by 10.1 cm Sodium Iodide Thallium activated [NaI(T1)] crystal was used for determining the gamma-ray spectrum. This instrument, along with all other survey and sampling devices, was housed in the mobile laboratory, a converted Winnebago Motor Home.

Smear Surveys

Dry smears were taken throughout the floor pad area of Building 421 and the outside wall of the adjacent Building 331. A standard smear is performed by applying moderate pressure by the tips of the first two fingers to the back of the filter paper. All smears were taken with No. 1 Whatman filter paper, 4.25 cm in diameter. Smears of 930 cm^2 were normally taken. If an area or object was found which had a higher than normal instrument background, a smear of 100 cm^2 was taken. A smear of 100 cm^2 was also taken if an area indicated excessive dirt loading. The smears were counted in groups of ten using a 10-Wire Flat Plate Gas Proportional Detector, developed at ANL, utilizing an Eberline Mini Scaler Model MS-2. At least one smear of each group was removed and counted in the more sensitive Nuclear Measurement Corporation 2π Internal Gas Flow Proportional Counter Model 5 (PC-5) using a mylar spun top. (The mylar spun top is placed over samples to negate the dielectric effect when counting samples on nonconducting media such as paper.) This procedure was used as a more sensitive means of counting a selection of the smear samples. All smears of areas or objects with elevated direct readings, were counted in the PC Counter. In addition, any smears indicating readings above the instrument background levels in the 10-Wire Assembly, were also counted in the PC-5 Counter. Smears were

counted in each detector for both alpha and beta activity. Appendix 1 provides the instrumentation and smear count conversion factors used.

Table 1 includes the survey results, while the map in Figure 1A indicates the location of the smears. A number, n, indicates the location of the particular smear. Figure 1B indicates the smear locations of the areas above background; R-1 through R-4 indicate the areas with an elevated direct reading.

Air Samples

Air samples were collected using a particulate air sampler (Princess Model) developed at ANL. (See Figure 1C for locations of air samples.) The collection medium consisted of a 200 cm^2 sheet of Hollingsworth-Vose (HV-70-0.23 mm) filter paper which collects the particulates present in the air. Flow rates of 15 or 40 cubic meters per hour (m^3/h) were used. A 10% portion, 5 cm in diameter, was removed from the filter media after collection and counted for both alpha and beta activity in the PC-5, utilizing a mylar spun top. The collection efficiency at these flow rates for 0.3 micron particles is approximately 99.9%. Sampling results were used to determine radon concentrations and the presence of any long-lived activity. These air sample calculations are presented in Appendix 2.

Soil Samples

Soil corings were taken at selected locations in undisturbed areas about the immediate periphery of Building 421 to determine the deposition, if any, of radionuclides that could have been spilled or released from the site. Figure 1D indicates the location of the soil samples taken at the Watertown Arsenal. Two additional sites were selected for background soil corings. These included a park in Newton, MA., and a triangular plot near a reservoir at

Stoneham, MA. (Figure 1E indicates the background soil sample locations.) Two corings were taken at each site to determine background levels in the area. This was also done to identify any spurious anomalies that can occur. Radiochemical (fluorometric) and gamma-ray spectral analyses were conducted on all soil samples.

The corings were effected using a 10.1 cm in diameter by 15.2 cm in length right circular cylinder cutting tool, used as a golf-green hole cutter. Each core was 30 cm in length and divided into four segments. Starting from the surface, three separate 5 cm segments are cut, bagged, and marked A, B, and C, respectively; the final segment of 15 cm was marked D. The reason for the segmented coring is to determine what, if any, contamination migration has occurred, to reduce the dilution of lower level soil with the upper level segments with respect to the surface deposition of the contaminants or vice versa, and to reveal any overburden or back fill that may have occurred over the years.

Eight soil corings were effected from the grounds adjacent to the Building 421 pad area. In addition, certain of these samples were chosen so as to be in the immediate vicinity of Building 34 and Building 41 sites; namely, samples S-4 and S-5 were taken just south of the old Building 34 site, and sample S-7 was taken just east of the old Building 41 site. Sample S-6 encompassed both the 421 and 41 sites since it was located between the two areas. Figure 1D depicts the soil sample locations.

All soil samples were processed at ANL (Figure 3) and shipped to a commercial laboratory (LFE Environmental Analysis Laboratories) for radiochemical (fluorometric) and gamma spectral analysis. Their soil analysis procedure is described in Appendix 3.

Sample preparation consisted of weighing the samples in their entirety and then drying for approximately 24 hours at 80° Centigrade. All samples were then reweighed, placed into mill jars (8.7ℓ), and milled until a sufficient amount of the soil sample could pass a No. 30 standard sieve. At no point were the rocks and heavy material ground or pulverized since this material would act as a diluent and, hence, lower the concentration of deposited material. After sufficient milling, the material was sieved using a No. 30 (600 micron) standard stainless sieve. The rocks and dross vs. sieved material were segregated, bagged, and weighed separately. Soil sample weights are given in Table 6. Aliquots of the sieved material were then loaded into screwtop plastic containers. The amount varied according to the type of analysis to be performed, 100 grams for gamma spectral and radiochemical (fluorometric) analysis and 10 grams for radiochemical (fluorometic) only. Every effort was made throughout the sample preparation operations to eliminate cross contamination. Soil samples which were suspected of containing elevated amounts of radioactivity were processed in equipment separate from the soil samples considered to contain background levels. All items of equipment were scrubbed and air dried prior to the introduction of the next sample.

Water Sample

A water sample was obtained from the storm sewer which ran along the north side of the Building 421 concrete pad. Since it was a single sewer line, only one sample was obtained. A 118 ml sample was taken at the location as depicted in Figure 1B. A Nalgene plastic bottle was taped to a long rod and was lowered into the sewer. The water sample was then placed on the Nuclear Data Multichannel Analyzer Model ND-100 to determine if any radioactivity was present.

ANALYSIS OF SURVEY RESULTS

General

This section discusses the results of the survey and the findings. The PAC-4G-3 instrument readings and smear results were normalized to units of disintegrations per minute per one hundred square centimeters (dis/min-100 cm^2) using the factors in Appendix 1 and are equated to natural uranium. All data is reported in net counts, i.e., the background count rates have been subtracted from the gross count rates prior to converting from counts per minute per one hundred square centimeters (c/min-100 cm^2) to dis/min-100 cm^2. The beta mode readings are corrected to remove any alpha contribution. The μR background readings which were taken for various lengths of time have been converted to μR per hour. Table 3 summarizes the average background reading for all modes of operation of the different instruments used. The μR readings taken about the pad are presented in Table 4 and their locations are depicted in Figure 1C.

The area accessible for survey on the floor pad was approximately 95% and all of the outside wall area of Building 331 up to 2 meters, was surveyed.

Instrument Surveys

There were four areas on the concrete floor pad that indicated above background readings, R-1, R-2, R-3 and R-4. The locations of these spots are shown in Figure 1B. The beta-gamma readings at these four locations were 8.4×10^2 dis/min-100 cm^2, 2.2×10^5 dis/min-100 cm^2, 2.2×10^5 dis/min-100 cm^2 and 8.5×10^4 dis/min-100 cm^2, respectively. No alpha contamination was found at these locations. Spots R-1 and R-3 were found in

cracks in the concrete pad. Spot R-2 was in a depression in the concrete. Spots R-1, R-2 and R-3 were 100 cm^2 or less. Spot R-4 was found on the flat surface of the concrete and was approximately 5 000 cm^2.

The highest End Window reading was indicated at Spot R-2. The End Window reading was 0.09 mR/h with the instrument in contact with the spot. None of the other three spots indicated an elevated direct reading with the End Window Detector.

Eight background measurements using a Micro R meter were taken on the building pad. These μR readings ranged from 4.9 μR/h to 11.5 μR/h. The range of μR background readings in an uncontaminated area is 7-12 μR/h. All of the μR readings taken on the concrete pad were found to be in this range.

A sample of spot R-2 was extracted and submitted for gamma spectral analysis. Figure 2 shows the gamma spectrum that was observed using the Multichannel Analyzer. Results of this analysis indicate the contaminant to be natural uranium; therefore, the PAC-4G-3 instrument readings are equated to natural uranium.

No elevated readings were found on the outside wall of Building 331.

Smear Surveys

Smears were randomly taken on the Building 421 floor pad and on the Building 331 wall which is next to the Building 421 floor pad. Smears were also taken on the four spots of contamination found on the floor pad. No smearable contamination was found.

Air Samples

The three air sampling results are presented in Table 5. The air sample locations are indicated as 1, 2 and 3 in Figure 1C. Using the calculations as

shown in Appendix 2, the Radon-222 Working Level (WL) concentrations were determined at these locations. The results were 0.00013 WL, 0.00011 WL and 0.00009 WL, respectively.

According to the Surgeon General's Guidelines found in 10 CFR 712, concentrations of radon daughters below 0.03 WL do not require remedial action in structures other than private dwellings or schools. (A copy of the Surgeon General's Guidelines is found in Appendix 6.)

Soil Samples

Results submitted by LFE Environmental Analysis Laboratories, as listed in Table 8, are reported in picocuries per gram (pCi/g) for the Germanium (Lithium) [Ge(Li)] spectral analysis and in micrograms per gram (μg/g) for the uranium fluorometric analysis. The latter concentrations were converted to pCi/g by means of the example calculation shown in Appendix 4.

The soil sample data along with background soil sample data is presented in Table 7. The background soil samples indicate natural uranium concentrations ranging from 0.9 to 4.8 pCi/g with one exception. This is SM-2B which was 12.2 pCi/g. Since samples SM-1 and SM-2 were taken in close proximity, it is expected that their results should be similar. The 12.2 pCi/g sample may be a result of soil fertilization. Since elevated concentrations of uranium are present where phosphate-containing fertilizers are used, any fertilizing of the soil can result in an increased uranium concentration. Results of soil samples taken at the Watertown Arsenal fall within the range of 0.9 to 4.8 pCi/g of uranium.

Water Sample

The water sample obtained in the storm sewer was counted on the Multichannel Analyzer. No radioactivity above background was detected in this sample.

Contamination Guidelines

Both the draft ANSI Standard N13.12 for the "Control of Radioactive Surface Contamination on Materials, Equipment, and Facilities to Be Released for Uncontrolled Use" and the NRC Guideline for "Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for By-Product, Source, or Special Nuclear Material", were considered as guidelines (see Appendix 5).

Since natural uranium was identified, the acceptable surface contamination levels for natural uranium will be used. The NRC Guideline for natural uranium is as follows: the average is 5 000 dis/min-100 cm^2 α, the maximum is 15 000 dis/min-100 cm^2 α, and the removable is 1 000 dis/min-100 cm^2 α. The measurements for the average may not be averaged over more than one square meter and the maximum level applies to an area of not more than 100 cm^2. Also, the average and maximum radiation levels associated with surface contamination resulting from beta-gamma emitters should not exceed 0.2 mrad/h at 1 cm and 1.0 mrad/h at 1 cm, respectively, measured through not more than 7 mg/cm^2 of total absorber.

The ANSI Standard N13.12 is more limiting than the NRC Guide for natural uranium. The allowable natural uranium activity is 5 000 dis/min-100 cm^2 total of which only 1 000 dis/min-100 cm^2 can be removable. The levels may be averaged over one square meter provided the maximum activity in

any area of 100 cm^2 is less than three times the limit value. Thus, the ANSI Standard is more restrictive in that its limits include both alpha and beta activity and the NRC Guide includes only alpha activity. In the areas of contamination found at Watertown Arsenal, beta-gamma activity is present, but no alpha activity was detectable. It is important to note that a thin layer of dirt on the concrete pad could prevent the detection of alpha particles. If this occurred, an area of contamination would meet the NRC Guideline, but would exceed the ANSI Standard levels.

The areas of fixed contamination which were found to be greater than the 5 000 dis/min-100 cm^2 total for natural uranium as given in the ANSI Standard are depicted in Figure 1B and are as follows:

Location	Counting Results
R-2	2.2×10^5 dis/min-100 cm^2
R-3	2.2×10^5 dis/min-100 cm^2
R-4	8.5×10^4 dis/min-100 cm^2.

HAZARD EVALUATION

In an effort to evaluate the radiological health hazard associated with the four contaminated spots on the pad at the Watertown Arsenal, a credible, but hypothetical situation will be determined. One such situation which could arise, would involve the disturbance of contaminated concrete. A pneumatic jackhammering of the contaminated spot could result in a radioactive aerosol. This would appear to be a satisfactory form to consider for purposes of assessing a potential radiological hazard.

Even though the contamination found at R-4 is not the highest per unit area, this spot will be used to assess the radiological hazard since it consists of the greatest level of total activity. The level of activity found at R-4 was 8.5×10^4 dis/min-100 cm^2 equated to natural uranium[1] ($dis/min_{(nu)}$-100 cm^2).

When the sample from R-2 was taken by scraping the concrete with the spatula, the level of activity was decreased by approximately one half. It would appear that most of the activity at R-2 is confined to the top 0.6 cm of the concrete. But since the contamination at R-4 is not found in cracks in the cement, it will be assumed that this level of 8.5×10^4 dis/min-100 cm^2 is constant through a depth of 10 cm and that this level of 8.5×10^4 dis/min-100 cm^2 is present in every 0.6 cm depth. This assumption will result in a conservative estimate of the internal exposure hazard.

Since the spot of contamination encompassed an area no larger than 5 000 cm^2 and the the depth of the jackhammering would be 10 cm, the total activity (A) involved would be as follows:

$$A = \frac{8.5 \times 10^4 \text{ dis/min}_{(nu)}\text{-}100 \text{ cm}^2}{0.6 \text{ cm}} \cdot 10 \text{ cm} \cdot 5\,000 \text{ cm}^2$$

$$= 7.1 \times 10^7 \text{ dis/min}_{(nu)}.$$

This is also equal to

$$A = 7.1 \times 10^7 \text{ dis/min}_{(nu)} \cdot \frac{1 \text{ } \mu\text{Ci}_{(nu)}}{2.22 \times 10^6 \text{ dis/min}_{(nu)}}$$

$$= 3.2 \times 10^1 \text{ } \mu\text{Ci}_{(nu)}.$$

Assuming the depth of the concrete cut from the jackhammer to be 10 cm, the volume of concrete displaced (B) would be as follows:

$$B = 10 \text{ cm} \cdot 5\,000 \text{ cm}^2 = 5 \times 10^4 \text{ cm}^3.$$

The activity per unit volume of concrete (C) would be as follows:

$$C = \frac{3.2 \times 10^1 \text{ } \mu\text{Ci}_{(nu)}}{5 \times 10^4 \text{ cm}^3} = \frac{6.4 \times 10^{-4} \text{ } \mu\text{Ci}_{(nu)}}{\text{cm}^3}.$$

Since jackhammering results mainly in large pieces of concrete rather than small particles, the assumption will be made that only 10% of the concrete will become airborne. Thus, the total activity that would become airborne is

3.2 $\mu Ci_{(nu)}$. The density of ordinary concrete is approximately 3 g/cm^3[2]. The activity per gram of total concrete suspended in the air (D) would then be:

$$D = \frac{6.4 \times 10^{-4}\ \mu Ci_{(nu)}}{cm^3} \cdot \frac{cm^3}{3g} = \frac{2.1 \times 10^{-4}\ \mu Ci_{(nu)}}{g}$$

One can assume that the concrete is dry and that a maximum dust burden would be created from the jackhammer disturbance.

This maximum dust burden would, however, most probably be no greater than the Threshold Limit Value (TLV) for Nuisance Particulates. The TLV is 15 mg/m^3 of air or 1.5×10^{-2} g/m^3[3].

If an aerosol was generated which contained this TLV, the concentration of the radionuclide (E) in the air would be:

$$E = \frac{2.1 \times 10^{-4}\ \mu Ci_{(nu)}}{g} \cdot \frac{1.5 \times 10^{-2}\ g}{m^3} = 3.2 \times 10^{-6}\ \mu Ci_{(nu)}/m^3$$

OR

$3.2 \times 10^{-12}\ \mu Ci_{(nu)}/cm^3$.

It should be realized that only a limited number of people could be directly involved in such a postulated environment for any length of time. In addition, the level of concrete dust described herein would probably be reduced at the height of a person's breathing zone due to the high density of the concrete. The Maximum Permissible Concentration in the air (MPCa) for natural uranium in an uncontrolled area is 2×10^{-12} $\mu Ci_{(nu)}/cc$[4].

Comparing the postulated level to the MPC, the following ratio (F) is obtained:

$$F = \frac{3.2 \times 10^{-12}\ \mu Ci/cm^2}{2 \times 10^{-12}\ \mu Ci/cc} = 1.6.$$

In the preceding situation we have generated an aerosol 1.6 times the MPCa.

Since the jackhammering of the small area would not take very long and the particles would soon fall out of suspension, the aerosol created is assumed to last only 10 minutes. A person involved in this operation for this period of time would inhale the following level of activity (G):

$$G = \frac{1.2\ m^3\ of\ air}{60\ min} \cdot 10\ min \cdot \frac{3.2 \times 10^{-6}\ \mu Ci_{(nu)}}{m^3} = 6.4 \times 10^{-7} \mu Ci_{(nu)}.$$

The fraction reaching the organ of reference, which in this case is the kidneys, would be 0.028[6]. Therefore, the total reaching the kidneys (H) would be:

$$H = 6.4 \times 10^{-7} \mu Ci_{(nu)} \cdot 0.028 = 1.8 \times 10^{-8} \mu Ci_{(nu)}.$$

The maximum permissible burden of uranium for the kidneys in the total body, [q(k)], is $5 \times 10^{-3}\ \mu Ci_{(nu)}$[7]. Comparing the total reaching the kidneys to the maximum permissible kidney burden the following value (I) is obtained:

$$I = \frac{1.8 \times 10^{-8}\ \mu Ci_{(nu)}}{5 \times 10^{-3}\ \mu Ci_{(nu)}} = 3.6 \times 10^{-6}.$$

Thus, the person would receive 3.6×10^{-6} of a kidney burden from this operation.

Even though these calculations are based on reasonable hypothesized values, it must be realized that the actual activity could be greater than the calculated value. Simultaneous use of the jackhammer at the other three contaminated areas would not significantly increase the hazard.

Only the GM End Window reading taken in contact with the spot R-2 was found to be greater than the instrument background. The reading at this spot was 0.09 mR/h. None of the GM End Window readings taken 1 meter from the contaminated spots were distinguishable from the normal instrument background range of 0.03 to 0.05 mR/h. Therefore, no external hazard is envisioned.

FOOTNOTES

1. A Curie of normal or natural uranium means the sum of 3.7×10^{10} dis/s from ^{238}U plus 3.7×10^{10} dis/s from ^{234}U plus 1.7×10^{9} dis/s from from ^{235}U. This equals 4.5×10^{12} dis/min per Curie of natural uranium. A standard Curie is 2.22×10^{12} dis/min.

2. Density of Elements and Common Material, as given in the Radiological Health Handbook, January, 1970.

3. The Threshold Limit Value is 15 mg/m^3 for Nuisance Particulates from the Occupational Safety and Health Standards subpart Z - Toxic and Hazardous Substances, May 28, 1975.

4. The Maximum Permissible Concentration values are given in "Standard for Protection Against Radiation," Code of Federal Regulations, Title 10, Part 20, Appendix B (April 30, 1975) (10 CFR 20), for an insoluble radionuclide in air in an uncontrolled area.

5. Specifications for Standard Man as given in the Radiological Health Handbook, January, 1970.

6. Report of Committee II of Permissible Dose from Internal Radiation (1959), International Committee on Radiological Protection (ICRP).

7. This is the maximum permissible kidney burden for ^{238}U as found in the Report of Committee II of Permissible Dose from Internal Radiation (1959), International Committee on Radiological Protection (ICRP).

SUMMARY OF FINDINGS

A comprehensive radiological survey was completed at the U.S. Arsenal Watertown, Watertown, Massachusetts. This site was used as a laboratory and office areas for the preparation and processing of uranium.

Direct instrument survey of the pad of Building 421 and the south exterior wall of Building 331 which is adjacent to the pad, indicate four small areas of contamination on the Building 421 pad. These four spots, marked R-1, R-2, R-3 and R-4 on the maps, consist of the following levels of contamination: 8.4×10^2 dis/min-100 cm^2 beta-gamma, 2.2×10^5 dis/min-100 cm^2 beta-gamma, 2.2×10^5 dis/min-100 cm^2 beta-gamma and 8.5×10^4 dis/min-100 cm^2 beta-gamma. No alpha contamination was detected. Gamma spectral analysis of a chip of contaminated concrete from spot R-2 indicates that the contaminant is natural uranium. Three of the spots, R-2, R-3 and R-4 exceed the allowable limits for natural uranium as found in the ANSI Standard No. N13.12 "Control of Radioactive Surface Contamination on Materials, Equipment, and Facilities to Be Released for Uncontrolled Use." Smears of these four areas indicate that the contamination is fixed and not removable. None of the four areas of contamination were found to be larger than 5 000 cm^2.

The highest End Window reading of 0.09 mR/h was indicated at Spot R-2. None of the other three spots indicated an elevated direct reading with the End Window Detector.

Integrated background measurements using the HPI 1010 were taken at seven location on the pad. These readings ranged from 4.9 μR/h to 11.5 μR/h. The normal range of readings for the μR Meter in an uncontaminated

area is 7-12 μR/h. None of the readings taken on the Building 421 pad were greater than the normal background.

The concentrations of radon daughters were determined at three locations on the Building 421 pad. These were 0.00013 WL, 0.00011 WL and 0.00009 WL. According to the Surgeon General's Guidelines found in 10 CFR 712, concentrations of radon daughters below 0.03 WL do not require remedial action in structures other than private dwellings and schools.

Eight soil samples were taken at locations about the site. To obtain background soil sample data, samples were collected in areas which appeared to be undisturbed since the MED/AEC era. An elevated level of uranium was noted in one background soil sample. This elevated uranium level is probably a result of the addition of phosphate fertilizer to the soil. Disregarding this sample, the background soil samples ranged from 0.9 to 4.8 pCi/g of natural uranium.

The levels of uranium found in the soil samples taken about the site ranged from 0.8 to 4.1 pCi/g. Results of the soil samples indicated no elevated levels above the natural background levels in the soil.

One water sample was obtained from the storm sewer line on the north side of the Building 421 pad. A gamma spectral analysis indicated no radioactivity in the sample. It is therefore felt that no contamination is present in this sewer.

FIGURE 1A

BUILDING 421 PAD

SMEAR LOCATIONS

ANL-HP DWG. NO. 78-3

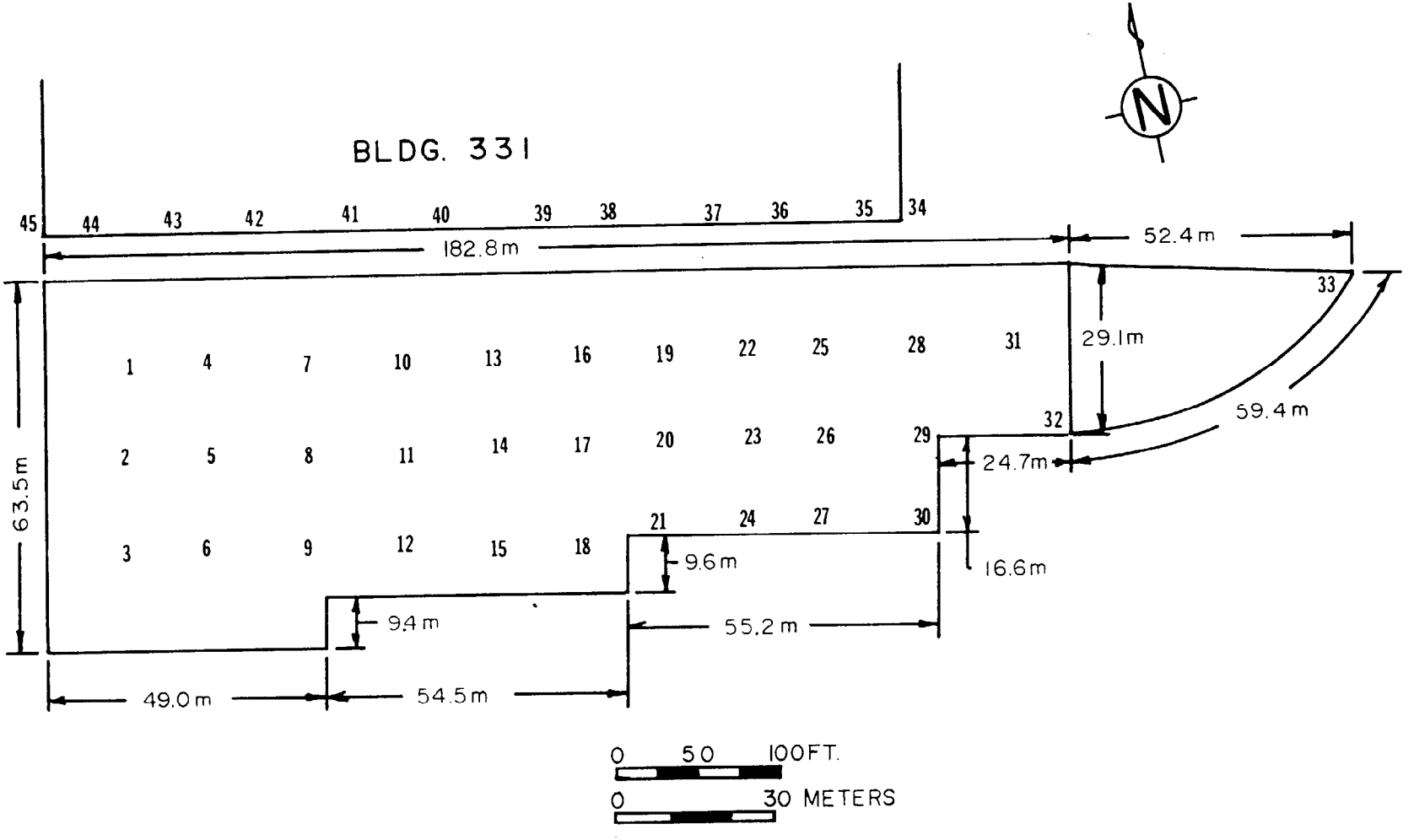


FIGURE 1B

CONTAMINATION AREAS ON BUILDING PAD

AND WATER SAMPLE LOCATION

ANL-HP DWG.NO. 78-4

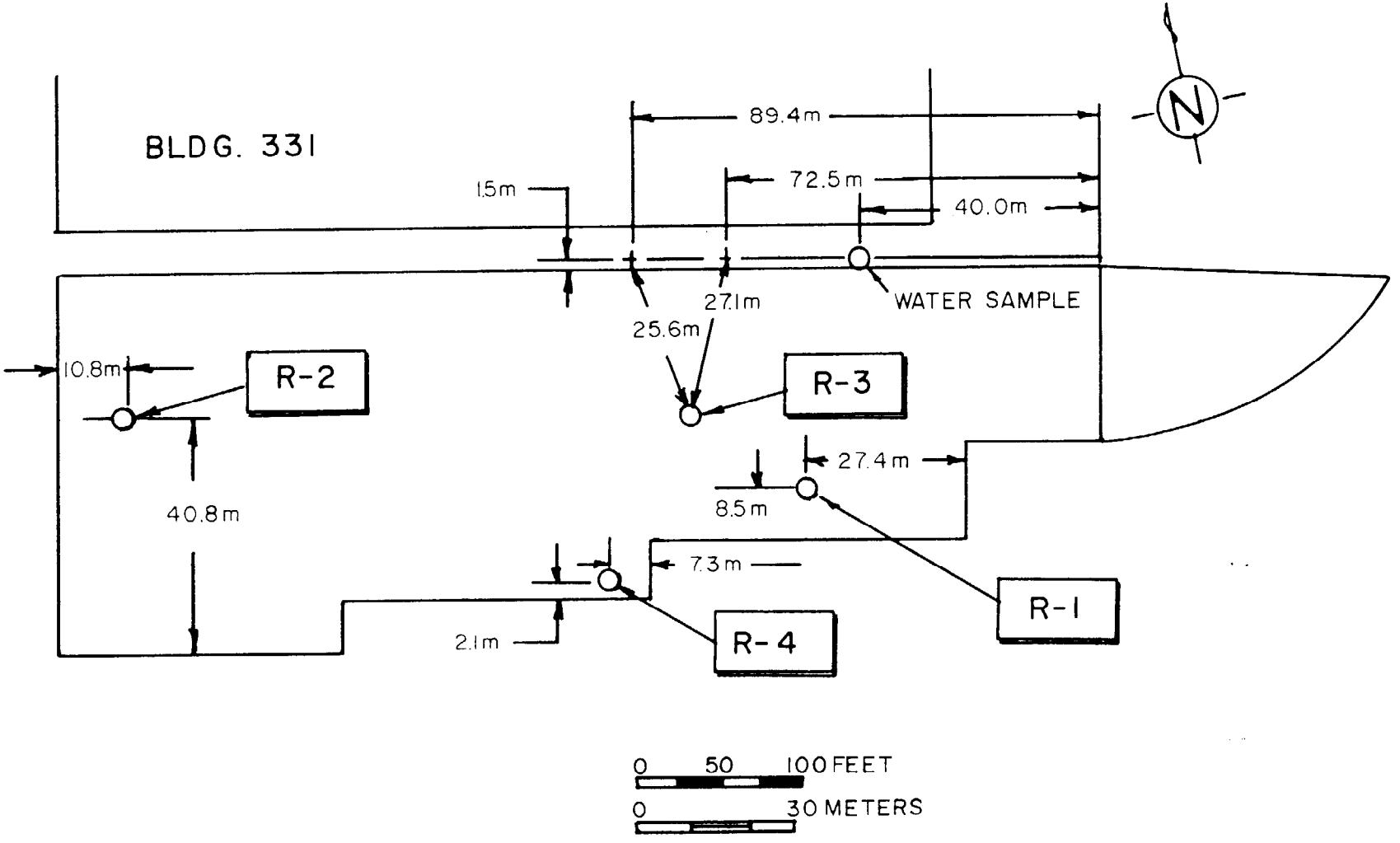


FIGURE 1C

MICRO R MEASUREMENTS AND AIR SAMPLE LOCATIONS ON BUILDING PAD

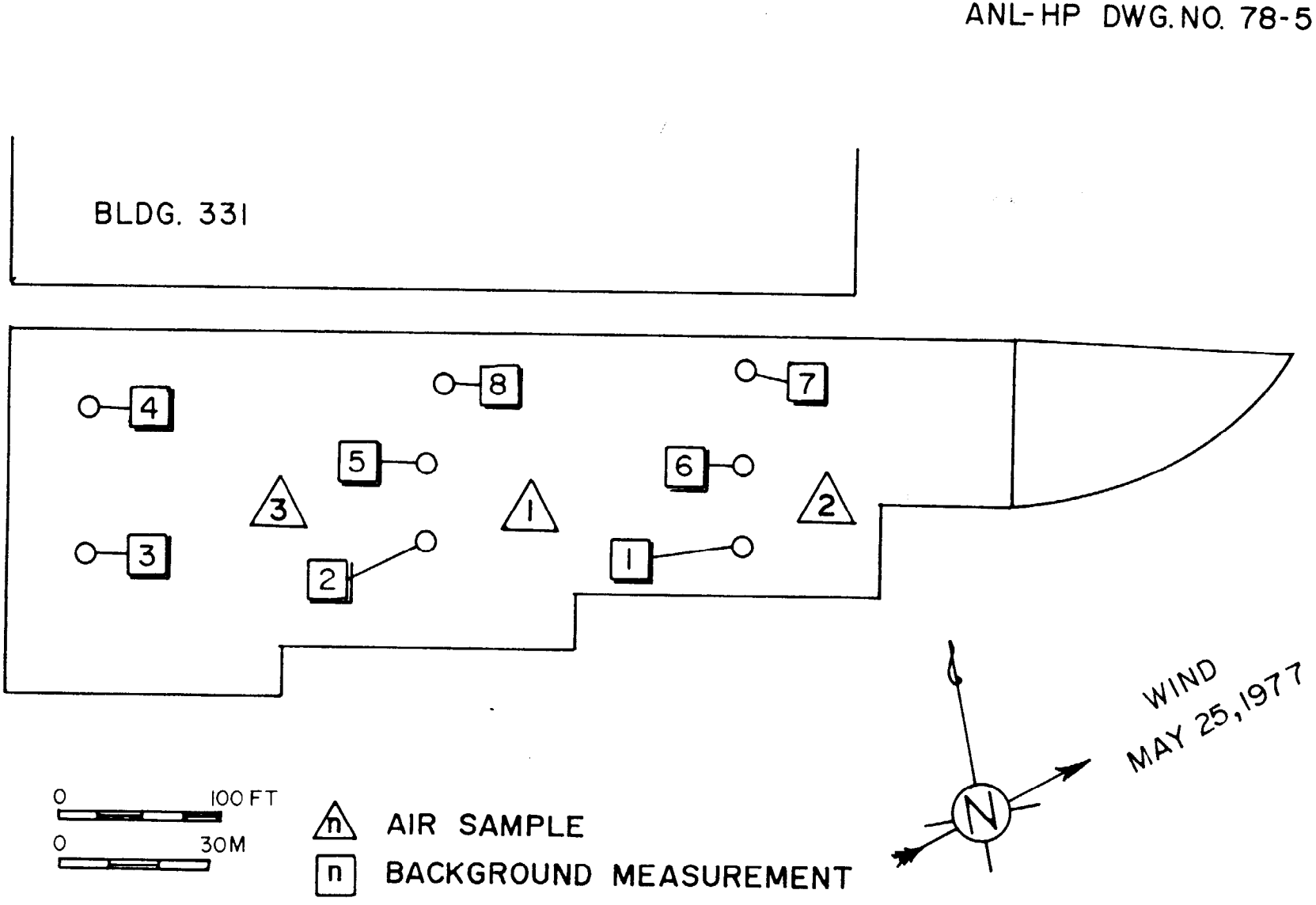


FIGURE 1D

SOIL SAMPLE LOCATIONS

ANL-HP DWG.NO. 78-6

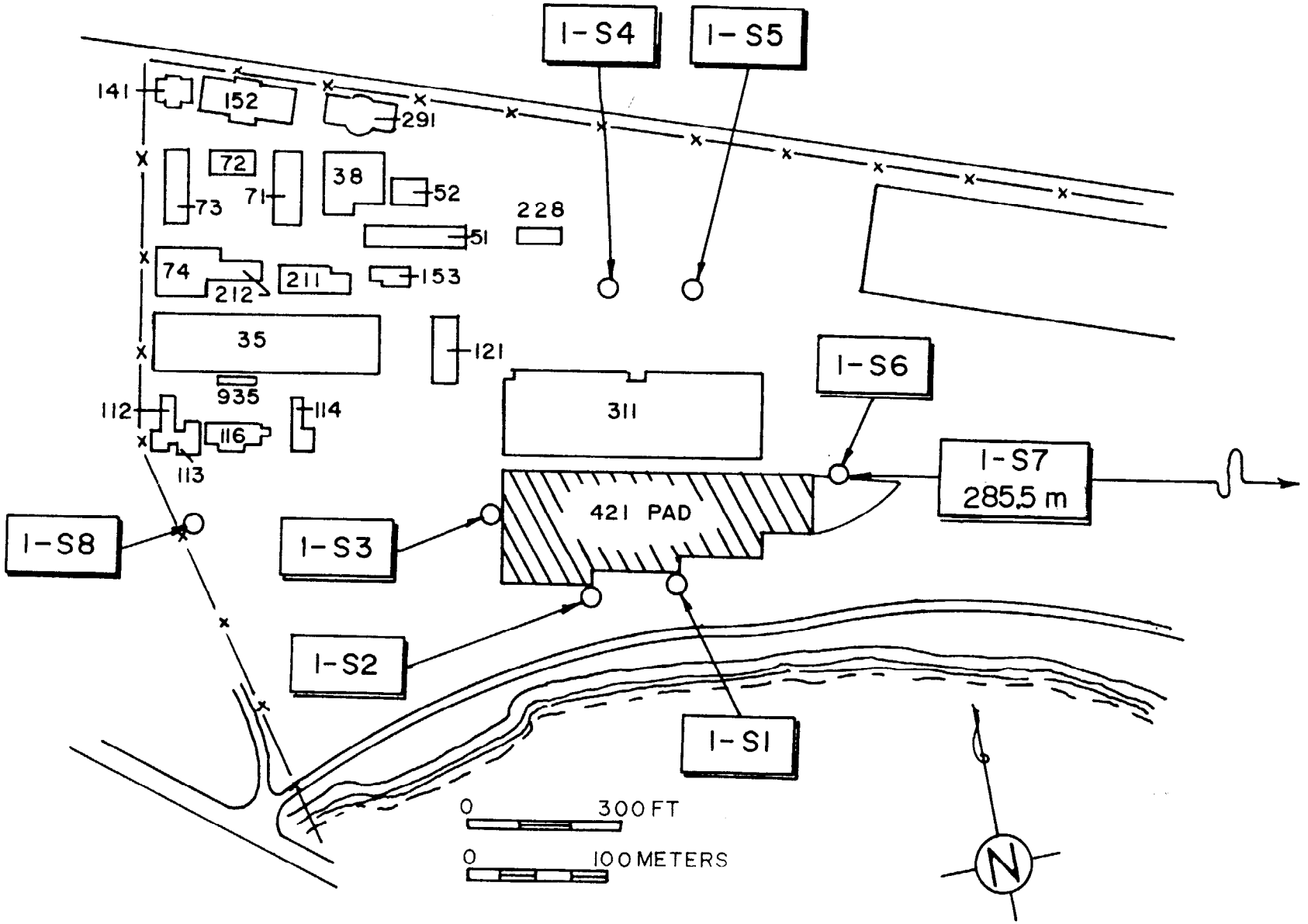


FIGURE 1E

BACKGROUND SOIL SAMPLE LOCATIONS

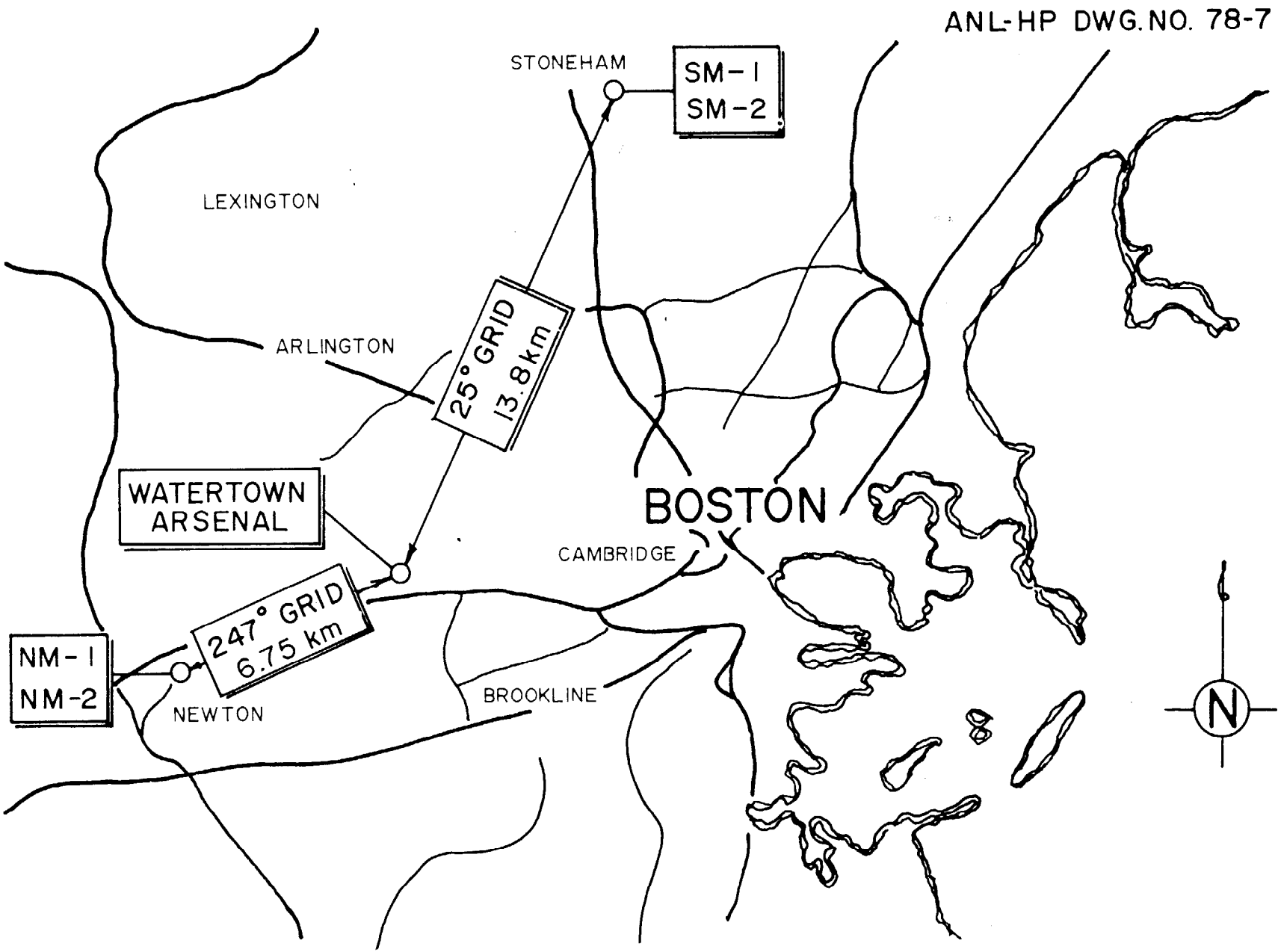


FIGURE 2
GAMMA SPECTRUM ANALYSIS OF CONTAMINATED SPOT ON PAD

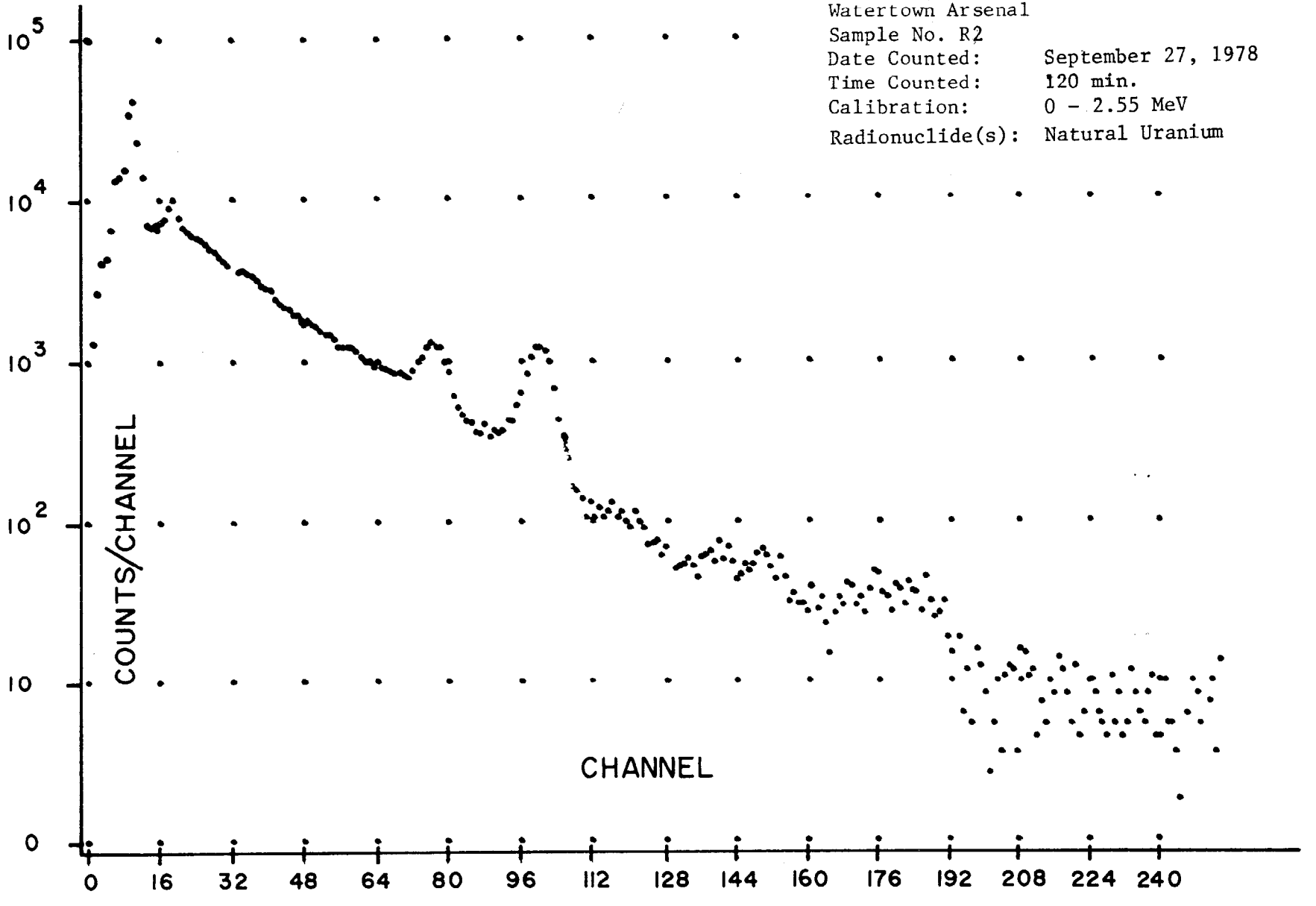
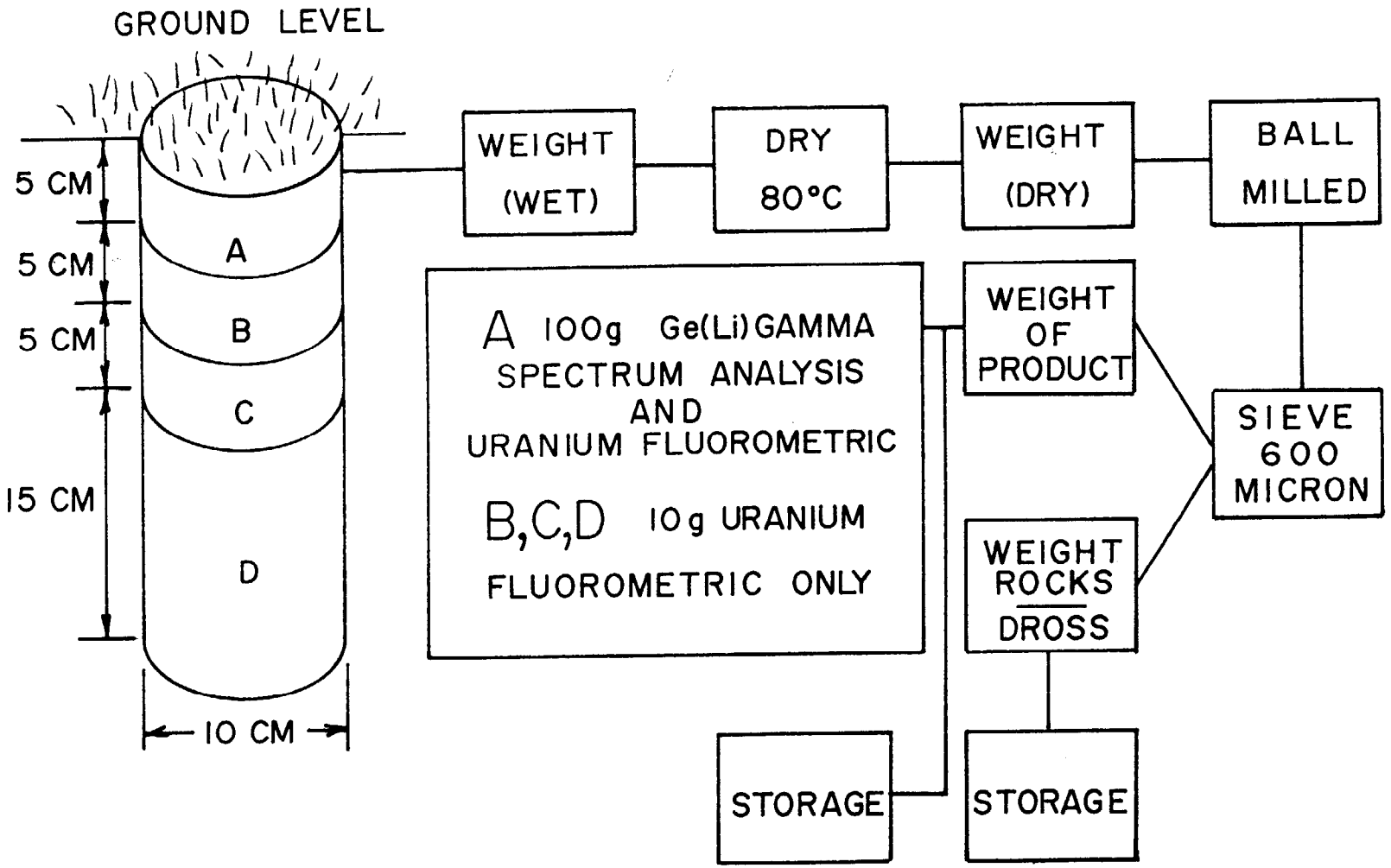


FIGURE 3

SOIL SAMPLING PROCEDURE AND PROCESSING DIAGRAM

ANL-HP-DWG. 78-2



COMMENTS ON TABLE 1

(a) The Beta Mode Direct Readings and Alpha Mode Direct Readings are taken with PAC-4G-3 instruments. The beta mode detects both electromagnetic and particulate radiation. If an area was found which indicated a higher count rate than the instrument background, a beta mode reading was obtained. The instrument was then switched to the alpha mode and a reading of the alpha contamination was obtained. In the alpha mode the instrument only responds to particles with high specific ionization such as alpha particles.

The beta mode readings are compensated for any alpha contribution.

(b) BKGD (Background) The following are the instrument background readings:

	Beta Mode	Alpha Mode
Floor Monitor	1500-2000 c/min-325 cm^2	0-50 c/min-325 cm^2
PAC-4G-3	150- 200 c/min-51 cm^2	0-50 c/min-51 cm^2
PC Counter	50 c/min	1.0 c/min
10 Wire	500 c/min	10 c/min

GM End Window Detector read 0.03 to 0.05 mR/h at 1 meter above floor.

(c) NA (Non-applicable) No contamination was detected above background in the beta mode; therefore, no alpha mode or contact GM End Window Survey was necessary.

TABLE I

DATA SHEET OF INSTRUMENT SURVEYS

Room No. or Area	Percent of Area Accessible for Survey		Air Sample (WL)	Direct Readings[a] (dis/min-100 cm^2)		End Window (mR/h)		Smear Results (dis/min-100 cm^2)	Comments
	Floors	Walls		Beta	Alpha	Contact	1 Meter		
Bldg.421 Foundation	95	No Walls	0.00013						Air Sample 1
			0.00011						Air Sample 2
			0.00009						Air Sample 3
				$8.4x10^2$	BKGD[b]	BKGD	BKGD	BKGD	Spot of contamination R-1 equated to natural uranium.
				$2.2x10^5$	BKGD	0.09	BKGD	BKGD	Spot of contamination R-2 equated to natural uranium.
				$2.2x10^5$	BKGD	BKGD	BKGD	BKGD	Spot of contamination R-3 equated to natural uranium.
				$8.5x10^4$	BKGD	BKGD	BKGD	BKGD	Spot of contamination R-4 equated to natural uranium.
				BKGD	NA[c]	NA	BKGD	BKGD	Rest of survey was BKGD.
Bldg.331	No floor survey	100		BKGD	NA	NA	BKGD	BKGD	Exterior of south wall ground level up to 2 meters.

TABLE 2

INSTRUMENTATION USED IN SURVEY

	Inventory Number	Probe Area	Window Thickness
Eberline Floor Monitor FM-4G utilizing a PAC-4G-3	181501	325 cm^2	0.85 mg/cm^2
Eberline Floor Monitor FM-4G utilizing a PAC-4G-3	183413	325 cm^2	0.85 mg/cm^2
Eberline Floor Monitor FM-4G utilizing a PAC-4G-3	184340	325 cm^2	0.85 mg/cm^2
PAC-4G-3	183412	51 cm^2	0.85 mg/cm^2
PAC-4G-3	183414	51 cm^2	0.85 mg/cm^2
PAC-4G-3	183415	51 cm^2	0.85 mg/cm^2
PAC-4G-3	184339	51 cm^2	0.85 mg/cm^2
PAC-4G-3	184341	51 cm^2	0.85 mg/cm^2
Eberline HP-90 Beta-Gamma End Window	159006	-	4.1 - 4.7 mg/cm^2
Nuclear Measurement Corp. PC-5 2π Internal Gas Flow Counter	184065	-	0.85 mg/cm^2
Argonne National Laboratory 10 Wire Flat Plate Gas Proportional Detector with Eberline Mini Scaler MS-2	184342 184343	400 cm^2	0.85 mg/cm^2
Argonne National Laboratory Filter Queen Air Sampler using HV-70 filter media	-	-	-
Health Physics Instrument Model HPI-1010	184444	-	-
Nuclear Data Multichannel Analyzer Model ND-100	184764	-	-

TABLE 3

INSTRUMENT BACKGROUND READINGS

Instrument	Readings* Alpha Mode (c/min)	Beta Mode (c/min)	1 meter above floor
Eberline Floor Monitor FM-4G using PAC-4G-3			
181501	0 - 50	1 500 - 2 000	
183413	0 - 50	1 500 - 2 000	
184340	0 - 50	1 500 - 2 000	
Eberline PAC-4G-3			
183412	0 - 50	150 - 200	
183414	0 - 50	150 - 200	
183415	0 - 50	150 - 200	
184339	0 - 50	150 - 200	
184341	0 - 50	150 - 200	
Eberline 530 with HP 190 Beta-Gamma End Window	-	-	0.03-0.05 mR/h
Nuclear Measurement Corp. PC-5 2π Internal Gas Flow Counter	0.3	40	
Argonne National Laboratory 10-Wire Flat Plate Gas Proportional Detector with Eberline Mini Scaler MS-2	10	500	
Health Physics Instrument Model HPI-1010			7-12 µR/h

*Background readings were initially taken in the mobile laboratory and rechecked throughout the various areas while surveying.

TABLE 4

MICRO R MEASUREMENTS TAKEN ON BUILDING 421 PAD

Number	Date	Reading* μR/h
1	5-23-77	11.5
2	5-23-77	10.0
3	5-23-77	8.2
4	5-24-77	5.3
5	5-24-77	10.9
6	5-24-77	9.7
7	5-25-77	9.4
8	5-25-77	4.9

*Normal background readings range from 7 - 12 μR/h.

TABLE 5

RADON CONCENTRATION DETERMINATIONS

Location	Date	dis/min-m^3	pCi/ℓ	WL*
1	5-25-78	29	0.013	0.00013
2	5-25-78	25	0.011	0.00011
3	5-25-78	20	0.009	0.00009

Example Calculation #1

$$0.013 \text{ pCi}/\ell \cdot \frac{\text{WL}}{100 \text{ pCi}/\ell} = 0.00013 \text{ WL}$$

*A Working Level (WL) is defined in 10 CFR 712 as any combination of short-lived radon daughter products in 1 liter of air that will result in the ultimate emission of 1.3×10^5 MeV of potential alpha energy. The numerical value of the WL is derived from the alpha energy released by the total decay through RaC' of the short-lived radon daughter products, RaA, RaB, RaC, at radioactive equilibrium with 100 pCi of ^{222}Rn per liter of air.

TABLE 6

SOIL SAMPLE WEIGHTS

Sample No.	Net Weight (grams)	Dry Weight (grams)	Sieved Weight (grams)	Rocks and Dross (grams)
		SOIL SAMPLES		
1-S1-A	466.1	451.4	405.1	39.6
1-S1-B	998.2	970.0	921.2	43.3
1-S1-C	823.6	791.2	754.1	34.5
1-S1-D	2219.0	2047.4	1910.6	128.2
1-S2-A	842.5	819.2	757.4	58.2
1-S2-B	685.5	650.8	604.4	45.4
1-S2-C	993.7	937.2	861.3	74.7
1-S2-D	2175.6	2032.4	1586.6	431.3
1-S3-A	906.0	887.9	840.0	33.5
1-S3-B	848.6	838.1	822.8	10.0
1-S3-C	1045.5	1022.9	942.4	75.7
1-S3-D	2749.0	2676.2	1875.5	785.2
1-S4-A	1051.9	1040.3	536.8	499.3
1-S4-B	1111.0	1093.1	418.9	666.1
1-S4-C	2326.7	2284.3	752.3	1526.1
1-S4-D	5208.6	5095.2	1416.4	3557.4
1-S5-A	781.9	777.8	569.8	201.7
1-S5-B	1192.2	1175.6	451.5	720.1
1-S5-C	1521.8	1494.2	701.2	781.7
1-S5-D	3058.8	2936.9	943.3	1961.5
1-S6-A	637.4	626.9	395.5	220.6
1-S6-B	851.0	826.6	481.1	337.7
1-S6-C	1048.3	1007.4	537.8	463.5
1-S6-D	2692.6	2553.8	926.3	1619.4
1-S7-A	241.1	234.1	197.4	26.8
1-S7-B	531.8	522.6	304.8	210.3
1-S7-C	1653.8	1631.0	551.4	1075.7
1-S7-D	5387.0	5347.2	1385.4	3915.0
1-S8-A	352.5	289.1	262.5	23.7
1-S8-B	368.4	308.4	299.3	3.5
1-S8-C	727.1	622.0	574.7	40.2
1-S8-D	1529.9	1522.8	942.6	350.4

TABLE 6

SOIL SAMPLE WEIGHTS
(cont'd)

Sample No.	Net Weight (grams)	Dry Weight (grams)	Sieved Weight (grams)	Rocks and Dross (grams)
		BACKGROUND SOIL SAMPLES		
NM-1-A	519.1	416.5	356.7	53.5
NM-1-B	668.4	588.8	334.0	247.5
NM-1-C	777.0	668.9	534.0	111.3
NM-1-D	1887.5	1622.9	1182.8	434.4
NM-2-A	752.5	613.9	471.7	131.7
NM-2-B	342.0	272.5	236.2	30.7
NM-2-C	983.0	814.3	699.1	108.3
NM-2-D	1779.4	1496.4	1271.3	211.4
SM-1-A	466.8	387.6	358.6	11.9
SM-1-B	789.9	705.4	425.4	275.0
SM-1-C	724.7	627.9	439.1	186.3
SM-1-D	1353.0	987.8	772.2	196.5
SM-2-A	811.1	666.6	612.2	49.5
SM-2-B	568.1	366.4	360.1	2.7
SM-2-C	487.8	331.7	324.0	3.5
SM-2-D	1214.4	849.6	835.0	10.6

TABLE 7

Ge(Li) SPECTRUM AND URANIUM FLUOROMETRIC ANALYSIS OF SOIL SAMPLES

Sample No.	Ge(Li) Spectra pCi/g received wt ± σ(1) ^{137}Cs	^{232}Th Decay Chain	^{226}Ra Decay Chain	Uranium μg/g ± σ(2)	Uranium pCi/g ± σ(3)
1-S1-A	0.09 ± 0.04	0.7 ± 0.1	0.61 ± 0.06	2.1 ± 0.4	1.5 ± 0.3
1-S1-B				1.5 ± 0.4	1.0 ± 0.3
1-S1-C				2.4 ± 0.9	1.7 ± 0.6
1-S1-D				1.8 ± 0.4	1.3 ± 0.3
1-S2-A	0.09 ± 0.04	0.6 ± 0.1	0.49 ± 0.06	1.9 ± 0.4	1.3 ± 0.3
1-S2-B				1.5 ± 0.4	1.0 ± 0.3
1-S2-C				1.5 ± 0.5	1.0 ± 0.3
1-S2-D				1.3 ± 0.3	0.9 ± 0.2
1-S3-A	0 ± 0.04	0.7 ± 0.1	0.51 ± 0.06	1.5 ± 0.4	1.0 ± 0.3
1-S3-B				1.5 ± 0.4	1.0 ± 0.3
1-S3-C				1.2 ± 0.3	0.8 ± 0.2
1-S3-D				1.2 ± 0.4	0.8 ± 0.3
1-S4-A	0.74 ± 0.04	0.8 ± 0.1	0.86 ± 0.06	2.2 ± 0.4	1.5 ± 0.3
1-S4-B				3.1 ± 0.6	2.2 ± 0.4
1-S4-C				3.1 ± 0.8	2.2 ± 0.6
1-S4-D				2.1 ± 0.6	1.5 ± 0.4
1-S5-A	2.6 ± 0.1	0.8 ± 0.1	0.86 ± 0.06	5.9 ± 0.7	4.1 ± 0.5
1-S5-B				2.5 ± 0.3	1.7 ± 0.2
1-S5-C				5.6 ± 0.7	3.9 ± 0.5
1-S5-D				5.0 ± 0.9	3.5 ± 0.6
1-S6-A	1.40 ± 0.07	0.9 ± 0.1	0.62 ± 0.06	3.1 ± 0.7	2.2 ± 0.5
1-S6-B				2.9 ± 1.4	2.0 ± 1.0
1-S6-C				2.5 ± 0.8	1.7 ± 0.6
1-S6-D				3.2 ± 0.8	2.2 ± 0.6
1-S7-A	2.7 ± 0.1	0.6 ± 0.1	0.62 ± 0.06	1.9 ± 0.7	1.3 ± 0.5
1-S7-B				2.2 ± 0.4	1.5 ± 0.3
1-S7-C				1.6 ± 0.4	1.1 ± 0.3
1-S7-D				2.5 ± 0.4	1.7 ± 0.3

TABLE 7

Ge(Li) SPECTRUM AND URANIUM FLUOROMETRIC ANALYSIS OF SOIL SAMPLES (cont'd)

Sample No.	Ge (Li) Spectra pCi/g received wt ± σ(1) ^{137}Cs	^{232}Th Decay Chain	^{226}Ra Decay Chain	Uranium μg/g ± σ(2)	Uranium pCi/g ± σ(3)
1-S8-A	3.3 ± 0.2	1.2 ± 0.2	1.58 ± 0.09	4.6 ± 0.5	3.2 ± 0.3
1-S8-B				2.8 ± 0.4	2.0 ± 0.3
1-S8-C				3.7 ± 0.5	2.6 ± 0.3
1-S8-D				3.8 ± 0.4	2.7 ± 0.3
		BACKGROUND SOIL SAMPLES			
NM-1-A	1.25 ± 0.06	1.2 ± 0.1	0.96 ± 0.06	3.4 ± 0.5	2.4 ± 0.3
NM-1-B				4.3 ± 0.6	3.0 ± 0.4
NM-1-C				4.1 ± 0.6	2.9 ± 0.4
NM-1-D				3.1 ± 0.4	2.2 ± 0.3
NM-2-A	0.91 ± 0.05	1.5 ± 0.1	1.00 ± 0.07	2.5 ± 0.5	1.7 ± 0.3
NM-2-B				6.8 ± 0.6	4.8 ± 0.4
NM-2-C				3.5 ± 0.5	2.4 ± 0.3
NM-2-D				3.2 ± 0.4	2.2 ± 0.3
SM-1-A	2.2 ± 0.1	0.9 ± 0.1	0.68 ± 0.06	2.5 ± 0.4	1.7 ± 0.3
SM-1-B				2.2 ± 0.4	1.5 ± 0.3
SM-1-C				1.3 ± 0.4	0.9 ± 0.3
SM-1-D				2.4 ± 0.4	1.7 ± 0.3
SM-2-A	0 ± 0.05	0.9 ± 0.1	0.72 ± 0.06	4.0 ± 0.4	2.8 ± 0.3
SM-2-B				17.4 ± 0.8	12.2 ± 0.6
SM-2-C				5.0 ± 0.4	3.5 ± 0.3
SM-2-D				3.2 ± 0.4	2.2 ± 0.3
LFE Blank	0 ± 0.4	0 ± 0.1	0 ± 0.06	0 ± 0.2	0 ± 0.1

(1) One standard deviation due to counting statistics.
(2) Data results from LFE.
(3) ANL conversion from Appendix 4.

APPENDIX 1

CONVERSION FACTORS

INSTRUMENTATION

Below are the conversion factors used to obtain the readings in units of disintegrations per minute per 100 cm^2 (dis/min-100 cm^2).

I Conversion Factors

	PAC-4G-3	Floor Monitor (FM-4G)
To 100 cm^2	1.96	0.31
c/min to dis/min (alpha)	2	2
c/min to dis/min (beta)	2	2
c/min to dis/min (alpha from natural uranium)	3.7	3.7
c/min to dis/min (beta-gamma from natural uranium)	4.3	3.6

II Deriviation of Conversion Factors

Floor Monitor (FM-4G)

Window Area: ~325 cm^2
Conversion to 100 cm^2 = 0.31 times Floor Monitor reading.

PAC-4G-3

Window Area: ~51 cm^2
Conversion to 100 cm^2 = 1.96 times PAC reading.

2π Internal Gas Flow Counter, PC-5

Geometry: Solid Stainless Steel Spun Top - 0.50
Geometry: Mylar Spun Top - 0.43
Mylar Spun Top Counting (double aluminized mylar window ~0.85 mg/cm^2) utilizes the well of the PC-5 and is a method developed and used by the Argonne National Laboratory Health Physics Section for negating the dielectric effect in counting samples on nonconducting media.

APPENDIX 1
(cont'd)

Utilizing a 3.2 cm x 3.2 cm x 0.3 cm natural uranium plate as a source of uranium alpha emissions, the plate was counted in the well of a 2π internal gas flow counter, PC-5, with the source leveled to an apparent 2π geometry.

The alpha c/min was found to be 27 596 c/min or 27 596/0.50 = 5.519×10^4 dis/min.

Using the floor monitor (FM-4G) and the uranium source to convert counts per minute (c/min) to disintegrations per minute (dis/min), the count was found to be 15 000 c/min at contact; therefore, $5.519 \times 10^4/15\ 000 = 3.7$ dis/min to c/min.

A similar reading was indicated on the PAC-4G-3, thus indicating the same factor for converting natural uranium c/min to dis/min from either the PAC-4G-3 or the floor monitor in the alpha mode.

The same natural uranium source, covered with 2 layers of nonconducting paper, each 6.65 mg/cm^2 to negate the alpha emissions, was counted for composite beta and gamma emissions in the PC-5. The source was leveled to an apparent 50% geometry; however, no provision was made for backscatter.

The composite beta-gamma count was found to be 538 066 c/min or 538 066/0.5 = 1.076×10^6 dis/min.

With the FM-4G floor monitor in the beta mode and in contact with the covered natural uranium source and centered on the probe, the count was found to be 300 000 c/min; this indicates a conversion factor of 1 076 000/ 300 000 = 3.6 dis/min to c/min.

Using the same covered source for the PAC-4G-3 beta mode conversion, in contact and centered on the probe, the count was found to be 250 000 c/min for a conversion factor of 1 076 000/250 000 = 4.3 dis/min to c/min.

SMEAR COUNT

The conversion factors for c/min-100 cm^2 to dis/min-100 cm^2 are given below.

I Conversion Equation (Alpha)

$$\frac{\text{c/min - (Bkgd)}}{\text{g} \cdot \text{bf} \cdot \text{sa} \cdot \text{waf}} = \text{dis/min Alpha}$$

A geometry (g) of 0.43 is standard for all flat plate counting.

A backscatter factor (bf) of 1.0 is used when determining alpha activity on a filter media.

APPENDIX 1
(cont'd)

The self-absorption (sa) was assumed to be 1 unless otherwise determined.

If the energies of the isotope were known, the appropriate window air factor (waf) was used; if the energies of the isotopes were unknown, the waf of ^{239}Pu, which is 0.713, was used.

The waf for natural uranium alphas is 0.54.

II Conversion Equation (Beta)

$$\frac{\text{c/min - [Beta Bkgd (c/min) + Alpha c/min]}}{\text{g} \cdot \text{bf} \cdot \text{sa} \cdot \text{waf}} = \text{dis/min Beta}$$

A geometry (g) of 0.43 is standard for all flat plate counting using the mylar spun top.

A backscattter factor (bf) of 1.1 is used when determining beta activity on a filter media.

The self-absorption (sa) was assumed to be 1 unless otherwise determined.

If the energies of the isotopes were known, the appropriate window air factor (waf) was used; if the energies of the isotopes were unknown, the waf of ^{90}Sr-^{90}Y, which is 0.85 was used.

The waf for natural uranium betas is 0.85.

APPENDIX 2

RADON DETERMINATION CALCULATIONS

This appendix summarizes the air sampling calculations for samples collected using an Argonne National Laboratory designed air sampler with HV-70 filter media. The attachment includes the basic assumptions and calculations used to derive the air concentrations.

I. Radon Concentrations Based on RaC' Results

The following postulates are assumed in deriving the Radon-222 (^{222}Rn) concentrations as based on the RaC' alpha count results.

1. RaA, RaB, RaC, RaC', are in equilibrium.

2. RaA is present only in the first count and not the 100 minute decay count.

3. One half of the radon progeny is not adhered to airborne particulates, and therefore, is not collected on the filter media.

4. The geometry factor (g) is 0.43 for both the alpha and beta activity.

5. The backscatter factor (bf) of 1.0 is used for the alpha activity which is determined from RaC'.

6. The sample absorption factor (sa) for RaC' is 0.77.

7. The window air factor (waf) for RaC' is 0.8.

8. RaB and RaC being beta emitters, are not counted in the alpha mode.

9. The half-life of the radon progeny is approximately 36 minutes, based on the combined RaB and RaC half-lives.

10. No long-lived alpha emitters are present as evidenced by the final count.

11. For all practical purposes, RaC' decays at the rate of the composite of RaB and RaC which is approximately 36 minutes.

APPENDIX 2
(cont'd)

II. Equations Used to Derive Air Concentrations

$$A_o = \frac{A}{e^{-\lambda t}}$$

Where: A_o = Activity present at the end of the sampling (dis/min)

A = Activity at some time interval, after end of sampling (dis/min)

t = Time interval from end of sampling period to counting interval

$$\lambda = \frac{0.693}{t_{\frac{1}{2}}}$$

$t_{\frac{1}{2}}$ = Half-life of isotope (min)

$$C = \frac{A_o \lambda}{f} \cdot \frac{1}{(1-e^{-\lambda t_s})}$$

Where: C = Concentration (dis/min-m^3)

A_o = Activity on filter media at end of sampling period (dis/min)

f = Sampling rate (m^3/min = m^3/h · 1h/60min)

t_s = Length of sampling time (min)

$$\lambda = \frac{0.693}{t_{\frac{1}{2}}}$$

$t_{\frac{1}{2}}$ = Half-life of isotope or controlling parent (min)

APPENDIX 2
(cont'd)

III. Example Calculation - No.1 as given in Table 5.

$$\dot{A}_o = \frac{15 \text{ dis/min}}{\exp \frac{-0.693 \cdot 100 \text{ min}}{36 \text{ min}}} = 102.8 \text{ dis/min}$$

$$C = \frac{102.8 \text{ dis/min} \cdot \frac{0.693}{36 \text{ min}}}{15/60 \text{ m}^3/\text{min}} \quad \frac{1}{1-\exp \frac{-0.693 \cdot 40 \text{ min}}{36 \text{ min}}}$$

$$= 14.7 \text{ dis/min-m}^3 \cdot 2 = 29 \text{ dis/min-m}^3$$

APPENDIX 3

SOIL ANALYSIS PROCEDURE FOR TOTAL URANIUM AND GAMMA-EMITTING NUCLIDES

Summary of Methods

A 60 milliliter volume of the received soil was counted in a petri dish for 500 minutes on a Ge(Li) detector over the energy range 0 - 1.5 MeV. This corresponded to between 60 to 100 g of soil, depending upon bulk soil density. Positive photopeaks above instrument background were converted to dis/min using a line efficiency curve based upon a National Bureau of Standards Multi Gamma standard. The natural Thorium-232 (^{232}Th) and Radium-226 (^{226}Ra) decay chains were calculated using the 0.910 MeV Actinium-228 (^{228}Ac) and 0.609 MeV Bismuth-214 (^{214}Bi) photopeaks respectively. Cesium-137 is reported for each sample as a representatitve gamma emitter. Potassium-40 (^{40}K) was observed on all soil samples, as expected, but was not calculated or reported.

One gram of the soil sample was ashed and dissolved in HF-HNO_3 for the total uranium analysis. A 100-λ aliquot of the dissolved sample was fused with 98% NaF-2% LiF and the fluorescence determined using a Jarrell-Ash fluorometer. A quenching factor was determined for each sample by using an internal spike.

APPENDIX 4

NATURAL URANIUM CALCULATIONS

Radioactive half-lives of ^{234}U, ^{235}U and ^{238}U, as well as the present abundance for each isotope were obtained as current best values from the "Table of Isotopes" - 6th Edition" by C. M. Lederer, J. M. Hollander and I. Perlman, 1967. The following values used are:

Isotope	Half-life (years)	% Abundance
^{234}U	2.47×10^5	0.0057
^{235}U	7.1×10^8	0.7196
^{238}U	4.51×10^9	99.2760
		100.0013

It should be noted that the abundance totals 100.0013%. Since it cannot be determined which isotope(s) are in error, the calculations are made with the 0.0013% error not accounted for.

$$SpA = \lambda N = \frac{(\ln 2)\, N}{t_{1/2}}$$

SpA = Specific Activity

Avogadro's Number = 6.025×10^{23}

N = Number of radioactive atoms per unit mass

$$= \frac{\text{Avogadro's Number}}{\text{gram atomic weight}}$$

$t_{1/2}$ = Half-life in years (a)

$$SpA = \frac{0.693 \cdot 6.025 \times 10^{23}}{t_{1/2}\,(a) \cdot 5.256 \times 10^5 \frac{\text{min}}{\text{a}} \cdot \text{gram atomic weight}} = \text{dis/min-gram}$$

$$SpA\ ^{234}U = \frac{0.693 \cdot 6.025 \times 10^{23}}{2.47 \times 10^5 \cdot 5.256 \times 10^5 \cdot 2.34 \times 10^2}$$

$$= 1.374 \times 10^{10} \text{ dis/min-gram}$$

$$= 1.374 \times 10^{4} \text{ dis/min-}\mu\text{gram} \cdot 5.70 \times 10^{-5}$$

$$= 0.783 \text{ dis/min-}\mu\text{gram of natural uranium}$$

APPENDIX 4
(cont'd)

$$\text{SpA } ^{235}U = \frac{0.693 \cdot 6.025 \times 10^{23}}{7.1 \times 10^{8} \cdot 5.256 \times 10^{5} \cdot 2.35 \times 10^{2}}$$

$$= 4.76 \times 10^{6} \text{ dis/min-gram}$$

$$= 4.76 \text{ dis/min-}\mu\text{gram} \cdot 7.196 \times 10^{-3}$$

$$= 0.034 \text{ dis/min-}\mu\text{gram of natural uranium}$$

$$\text{SpA } ^{238}U = \frac{0.693 \cdot 6.025 \times 10^{23}}{4.51 \times 10^{9} \cdot 5.256 \times 10^{5} \cdot 2.38 \times 10^{2}}$$

$$= 7.4 \times 10^{5} \text{ dis/min-gram}$$

$$= 0.74 \text{ dis/min-}\mu\text{gram} \cdot 9.9276 \times 10^{-1}$$

$$= 0.735 \text{ dis/min-}\mu\text{gram of natural uranium}$$

Therefore, the activity of 1 μgram of natural uranium is

$$0.783 \text{ dis/min } ^{234}U + 0.034 \text{ dis/min } ^{235}U + 0.735 \text{ dis/min}$$

$$= 1.552 \text{ dis/min-}\mu\text{gram}$$

$$= \frac{1.552 \text{ dis/min-}\mu\text{gram}}{2.22 \text{ dis/min-pCi}}$$

$$= 0.6991 \text{ pCi/}\mu\text{gram natural uranium}$$

Conversion of μg/g to pCi/g

Example Calculation - 1-S2-A as given in Table 7.

$$[1.9 \pm 0.4] \frac{\mu\text{gram}}{\text{gram}} \cdot \frac{0.6991 \text{ pCi}}{\text{gram}} = [1.3 \pm 0.3] \frac{\text{pCi}}{\text{gram}}$$

APPENDIX 5

PERTINENT RADIOLOGICAL REGULATIONS, STANDARDS AND GUIDELINES

APPENDIX 5
(cont'd)

Excerpts From

DRAFT AMERICAN NATIONAL STANDARD

N13.12

Control of Radioactive Surface Contamination on Materials, Equipment, and Facilities to be Released for Uncontrolled Use

APPENDIX 5
(cont'd)

Where potentially contaminated surfaces are not accessible for measurement (as in some pipes, drains, and ductwork), such property shall not be released pursuant to this standard, but shall be made the subject of case-by-case evaluation.

Property shall not be released for uncontrolled use unless measurements show the total and removable contamination levels to be no greater than the values in Table 1 or Table 2. (The values in Table 2 are easier to apply when the contaminants cannot be individually identified.)

Coatings used to cover the contamination shall not be considered a solution to the contamination problem. That is, the monitoring techniques shall be sufficient to determine, and such determination shall be made, that the total amount of contamination present on and under any coating does not exceed the Table 1 or Table 2 values before release.

APPENDIX 5
(cont'd)

TABLE 1
SURFACE CONTAMINATION LIMITS*

Group	Description	Nuclides (Note 1)	Limit (Activity) (dis/min-100 cm^2)+ Removable	Total (Fixed plus Removable)
1	Nuclides for which the non-occupational MPC_a (Note 2) is $2x10^{-13}$ Ci/m^3 or less or for which the non-occupational MPC_w (Note 4) is $2x10^{-7}$ Ci/m^3 or less	^{227}Ac, $^{241,242m,243}Am$, $^{249,250,251,252}Cf$, $^{243,244,245,246,247,248}Cm$, $^{125,129}I$, ^{237}Np, ^{231}Pa, ^{210}Pb, $^{238,239,240,242,244}Pu$, $^{226,228}Ra$, $^{228,230}Th$	20	Nondetectable (Note 3)
2	Those nuclides not in Group 1 for which the non-occupational MPC_a (Note 2) is $1x10^{-12}$ Ci/m^3 or less or for which the non-occupational MPC_w (Note 4) is $1x10^{-6}$ Ci/m^3 or less	^{254}Es, ^{256}Fm, $^{126,131,133}I$, ^{210}Po, ^{223}Ra, ^{90}Sr, ^{232}Th, ^{232}U	200	2 000 α, Nondetectable β (Note 5)
3	Those nuclides not in Group 1 or Group 2		1 000	5 000

APPENDIX 5

SURFACE CONTAMINATION LIMITS* (cont'd)

*The levels may be averaged over one square meter provided the maximum activity in any area of 100 cm^2 is less than three times the limit value. For purposes of averaging with regard to isolated spots of activity, any square meter of surface shall be considered to be contaminated above the limit L, applicable to 100 cm^2, if (1) from measurements of a representative number n of sections it is determined that $1/\eta \Sigma_n S_i > L$, where S_i is the dis/min-100 cm^2 determined from measurement of section i; or (2) it is determined that the activity of all isolated spots or particles in any area less than 100 cm^2 exceeds 3L.

+Disintegrations per minute per square decimeter.

NOTES:
(1) Values presented here are obtained from the Code of Federal Regulations, Title 10, Part 20, April 30, 1975. The most limiting of all given MPC values (for example, soluble versus insoluble) are to be used. In the event of the occurrence of mixtures of radionuclides, the fraction contributed by each constituent of its own limit shall be determined and the sum of the fractions shall be less than 1.

(2) Maximum permissible concentration in air applicable to continuous exposure of members of the public as published by or derived from an authoritative source such as the National Committee on Radiation Protection and Measurements (NCRP), the International Commission on Radiological Protection (ICRP), or the Nuclear Regulatory Commission (NRC). From the Code of Federal Regulations, Title 10, Part 20, Appendix B, Table 2, Column 1.

(3) The instrument utilized for this measurement shall be calibrated to measure at least 100 pCi of any Group-1 contaminants uniformly spread over 100 cm^2.

(4) Maximum permissible concentration in water applicable to members of the public.

(5) The instrument utilized for this measurement shall be calibrated to measure at least 1 nCi of any Group-2 beta or gamma contaminants uniformly spread over an area equivalent to the sensitive area of the detector. Direct survey for unconditional release should be performed in areas where the background is < 100 counts per minute. When the survey must be performed in a background exceeding 100 counts per minute, it may be necessary to use the indirect survey method to provide the additional sensitivity required.

APPENDIX 5
(cont'd)

TABLE 2
ALTERNATE SURFACE CONTAMINATION LIMITS

(all Alpha Emitters, except U nat and Th nat, Considered as a Group)*

Contamination Contigencies	Limit (Activity) (dis/min-100 cm^2) + Removable	Total (Fixed plus Removable)
If the contaminant cannot be identified; or if alpha emitters other than U nat (Note 1) and Th nat are present; or if the beta emitters comprise ^{227}Ac or ^{228}Ra	20	Nondetectable (Note 2)
If it is known that all alpha emitters are generated from U nat (Note 1) and Th nat; and if beta emitters are present that, while not identified, do not include ^{227}Ac, ^{125}I, ^{226}Ra, and ^{228}Ra	200	2 000 α Nondetectable β γ (Note 3)
If it is known that alpha emitters are generated only from U nat (Note 1) and Th nat in equilibrium with its decay products; and if the beta emitters, while not identified, do not include ^{227}Ac, ^{125}I, ^{129}I, ^{90}Sr, ^{223}Ra, ^{228}Ra, ^{126}I, ^{131}I and ^{133}I	1 000	5 000

APPENDIX 5

ALTERNATE SURFACE CONTAMINATION LIMITS (cont'd)

*The levels may be averaged over one square meter provided the maximum activity in any area of 100 cm^2 is less than three times the limit value. For purposes of averaging with regard to isolated spots of activity, any square meter of surface shall be considered to be contaminated above the limit L, applicable to 100 cm^2, if (1) from measurements of a representative number n of sections it is determined that $1/\eta \Sigma_{\eta} S_i > L$, where S_i is the dis/min-100 cm^2 determined from measurement of section i; or (2) it is determined that the activity of all isolated spots or particles in any area less than 100 cm^2 exceeds 3L.

+Disintegrations per minute per square decimeter.

NOTES:

(1) U nat and decay products.

(2) The instrument utilized for this measurement shall be calibrated to measure at least 100 pCi of any Group-1 contaminants uniformly spread over 100 cm^2.

(3) The instrument utilized for this measurement shall be calibrated to measure at least 1 nCi of any Group-2 beta or gamma contaminants uniformly spread over an area equivalent to the sensitive area of the detector. Direct survey for unconditional release should be performed in areas where the background is < 100 counts per minute. When the survey must be performed in a background exceeding 100 counts per minute, it may be necessary to use the indirect survey method to provide the additional sensitivity required.

APPENDIX 5
(cont'd)

GUIDELINES FOR DECONTAMINATION OF FACILITIES AND EQUIPMENT PRIOR TO RELEASE FOR UNRESTRICTED USE OR TERMINATION OF LICENSES FOR BY-PRODUCT, SOURCE, OR SPECIAL NUCLEAR MATERIAL

U. S. Nuclear Regulatory Commission
Division of Fuel Cycle and
Material Safety
Washington, D.C. 20555

November 1976

APPENDIX 5
(cont'd)

The instructions in this guide in conjunction with Table I specify the radioactivity and radiation exposure rate limits which should be used in accomplishing the decontamination and survey of surfaces or premises and equipment prior to abandonment or release for unrestricted use. The limits in Table I do not apply to premises, equipment, or scrap containing induced radioactivity for which the radiological considerations pertinent to their use may be different. The release of such facilities or items from regulatory control will be considered on a case-by-case basis.

1. The licensee shall make a reasonable effort to eliminate residual contamination.

2. Radioactivity on equipment or surfaces shall not be covered by paint, plating, or other covering material unless contamination levels, as determined by a survey and documented, are below the limits specified in Table I prior to applying the covering. A reasonable effort must be made to minimize the contamination prior to use of any covering.

3. The radioactivity on the interior surfaces of pipes, drain lines, duct work shall be determined by making measurements at all traps, and other appropriate access points, provided that contamination at these locations is likely to be representative of contamination on the interior of the pipes, drain lines, or duct work. Surfaces of premises, equipment, or scrap which are likely to be contaminated but are of such size, construction, or location as to make the surface inaccessible for purposes of measurement shall be presumed to be contaminated in excess of the limits.

4. Upon request, the Commission may authorize a licensee to relinquish possession or control of premises, equipment, or scrap having surfaces contaminated with materials in excess of the limits specified. This may include, but would not be limited to, special circumstances such as razing of buildings, transfer of premises to another organization continuing work with radioactive materials, or conversion of facilities to a long-term storage or standby status. Such request must:

   a. Provide detailed, specific information describing the premises, equipment or scrap, radioactive contaminants, and the nature, extent, and degree of residual surface contamination.

APPENDIX 5
(cont'd)

b. Provide a detailed health and safety analysis which reflects that the residual amounts of materials on surface areas, together with other considerations such as prospective use of the premises, equipment or scrap, are unlikely to result in an unreasonable risk to the health and safety of the public.

5. Prior to release of premises for unrestricted use, the licensee shall make a comprehensive radiation survey which establishes that contamination is within the limits specified in Table 1. A copy of the survey report shall be filed with the Division of Fuel Cycle and Material Safety, USNRC, Washington, D.C. 20555, and also the Director of the Regional Office of the Office of Inspection and Enforcement, USNRC, having jurisdiction. The report should be filed at least 30 days prior to the planned date of abandonment. The survey report shall:

   a. Identify the premises.

   b. Show that reasonable effort has been made to eliminate residual contamination.

   c. Describe the scope of the survey and general procedures followed.

   d. State the findings of the survey in units specified in the instruction.

Following review of the report, the NRC will consider visiting the facilities to confirm the survey.

APPENDIX 5
(cont'd)

TABLE 1

ACCEPTABLE SURFACE CONTAMINATION LEVELS

NUCLIDES[a]	AVERAGE[bcf]	MAXIMUM[bdf]	REMOVABLE[bef]
U-nat, U-235, U-238, and associated decay products	5 000 dis/min α-100 cm^2	15 000 dis/min α-100 cm^2	1 000 dis/min α-100 cm^2
Transuranics, Ra-226 Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	100 dis/min-100 cm^2	300 dis/min-100 cm^2	20 dis/min-100 cm^2
Th-nat, Th-232, Sr-90 Ra-223, Ra-224, U-232, I-126, I-131, I-133	1 000 dis/min-100 cm^2	3 000 dis/min-100 cm^2	200 dis/min-100 cm^2
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above.	5 000 dis/min $\beta\gamma$-100 cm^2	15 000 dis/min $\beta\gamma$-100 cm^2	1 000 dis/min $\beta\gamma$-100 cm^2

[a] Where surface contamination by both alpha- and beta-gamma-emitting nuclides exists, the limits established for alpha- and beta-gamma-emitting nuclides should apply independently.

[b] As used in this table, dis/min (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

APPENDIX 5

ACCEPTABLE SURFACE CONTAMINATION LEVELS
(cont'd)

[c]Measurements of average contaminant should not be averaged over more than 1 square meter. For objects of less surface area, the average should be derived for each such object.

[d]The maximum contamination level applies to an area of not more than 100 cm^2.

[e]The amount of removable radioactive material per 100 cm^2 of surface area should be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped.

[f]The average and maximum radiation levels associated with surface contamination resulting from beta-gamma emitters should not exceed 0.2 mrad/h at 1 cm and 1.0 mrad/h at 1 cm, respectively, measured through not more than 7 milligrams per square centimeter of total absorber.

APPENDIX 6

SURGEON GENERAL'S GUIDELINES

Part 712

Grand Junction Remedial Action Criteria

APPENDIX 6
(cont'd)

SURGEON GENERAL'S GUIDELINES
Part 712
Grand Junction Remedial Action Criteria

Federal Register, Vol.41, No.253, pp. 56777
Thursday, December 30, 1976

PART 712 GRAND JUNCTION
REMEDIAL ACTION CRITERIA

712.1 Purpose.

(a) The regulations in this part establish the criteria for determination by ERDA of the need for, priority of and selection of appropriate remedial action to limit the exposure of individuals in the area of Grand Junction, Colo., to radiation emanating from uranium mill tailing which have been used as construction-related material.

(b) The regulations in this part are issued pursuant to Pub. L. 92-314 (86 Stat. 222) of June 16, 1972.

712.2 Scope.

The regulations in this part apply to all structures in the area of Grand Junction, Colo., under or adjacent to which uranium mill tailings have been used as a construction-related material between January 1, 1951, and June 16, 1972, inclusive.

712.3 Definitions.

As used in this part:

(a) "Administrator" means the Administrator of Energy Research and Development or his duly authorized representative.

(b) "Area of Grand Junction, Colo.," means Mesa County, Colo.

(c) "Background" means radiation arising from cosmic rays and radioactive material other than uranium mill tailings.

APPENDIX 6
(cont'd)

(d) "ERDA" means the U.S. Energy Research and Development Administration or any duly authorized representative thereof.

(e) "Construction-related material" means any material used in the construction of a structure.

(f) "External gamma radiation level" means the average gamma radiation exposure rate for the habitable area of a structure as measured near floor level.

(g) "Indoor radon daughter concentration level" means that concentration of radon daughters determined by: (1) Averaging the results of 6 air samples each of at least 100 hours duration, and taken at a minimum of 4-week intervals throughout the year in a habitable area of a structure, or (2) utilizing some other procedure approved by the Commission.

(h) "Milliroentgen" (mR) means a unit equal to one-thousandth (1/1 000) of a roentgen which roentgen is defined as an exposure dose of X or gamma radiation such that the associated corpuscular emission per 0.001293 gram of air produces, in air, ions carrying one electrostatic unit of quantity of electricity of either sign.

(i) "Radiation" means the electromagnetic energy (gamma) and the particulate radiation (alpha and beta) which emanate from the radioactive decay of radium and its daughter products.

(j) "Radon daughters" means the consecutive decay products of radon-222. Generally, these include Radium A (polonium-218), Radium B (lead-214), Radium C (bismuth-214), and Radium C' (polonium-214).

APPENDIX 6
(cont'd)

(k) "Remedial action" means any action taken with a reasonable expectation of reducing the radiation exposure resulting from uranium mill tailings which have been used as construction-related material in and around structures in the area of Grand Junction, Colo.

(l) "Surgeon General's guidelines" means radiation guidelines related to uranium mill tailings prepared and released by the Office of the U.S. Surgeon General, Department of Health, Education and Welfare on July 27, 1970.

(m) "Uranium mill tailings" means tailings from a uranium milling operation involved in the Federal uranium procurement program.

(n) "Working Level" (WL) means any combination of short-lived radon daughter products in 1 liter of air that will result in the ultimate emission of 1.3×10^5 MeV of potential alpha energy.

712.4 Interpretations.

Except as specifically authorized by the Administrator in writing, no interpretation of the meaning of the regulations in this part by an officer or employee of ERDA other than a written interpretation by the General Counsel will be recognized to be binding upon ERDA.

712.5 Communications.

Except where otherwise specified in this part, all communications concerning the regulations in this part should be addressed to the Director, Division of Safety, Standards, and Compliance, U.S. Energy Research and Development Administration, Washington, D.C. 20545.

712.6 General radiation exposure level criteria for remedial action.

APPENDIX 6
(cont'd)

The basis for undertaking remedial action shall be the applicable guidelines published by the Surgeon General of the United States. These guidelines recommend the following graded action levels for remedial action in terms of external gamma radiation level (EGR) and indoor radon daughter concentration level (RDC) above background found within dwellings constructed on or with uranium mill tailings:

EGR	RDC	Recommendation
Greater than 0.1 mR/h	Greater than 0.05 WL	Remedial action indicated
From 0.05 to 0.1 mR/h	From 0.01 to 0.05 WL	Remedial action may be suggested
Less than 0.05 mR/h	Less than 0.01 WL	No remedial action indicated

712.7 Criteria for determination of possible need for remedial action.

Once it is determined that a possible need for remedial action exists, the record owner of a structure shall be notified of that structure's eligibility for an engineering assessment to confirm the need for remedial action and to ascertain the most appropriate remedial measure, if any. A determination of possible need will be made if as a result of the presence of uranium mill tailings under of adjacent to the structure, one of the following criteria is met:

APPENDIX 6

(cont'd)

(a) Where ERDA approved data on indoor radon daughter concentration levels are available:

(1) For dwellings and schoolrooms: An indoor radon daughter concentration level of 0.01 WL or greater above background.

(2) For other structures: An indoor radon daughter concentration level of 0.03 WL or greater above background.

(b) Where ERDA approved data on indoor radon daughter concentration levels are not available:

(1) For dwellings and schoolrooms:

(i) An external gamma radiation level of 0.05 mR/h or greater above background.

(ii) An indoor radon daughter concentration level of 0.0l WL or greater above background (presumed).

(A) It may be presumed that if the external gamma radiation level is equal to or exceeds 0.02 mR/h above background, the indoor radon daughter concentration level equals or exceeds 0.01 WL above background.

(B) It should be presumed that if the external gamma radiation level is less than 0.001 mR/h above background, the indoor radon daughter concentration level is less than 0.01 WL above background, and no possible need for remedial action exists.

(C) If the external gamma radiation level is equal to or greater than 0.001 mR/h above background but is less than 0.02 mR/h above background, measurements will be required to ascertain the indoor radon daughter concentration level.

APPENDIX 6
(cont'd)

(2) For other structures:

(i) An external gamma radiation level of 0.15 mR/h above background averaged on a room-by-room basis.

(ii) No presumptions shall be made on the external gamma radiation level/indoor radon daughter concentration level relationship. Decisions will be made in individual cases based upon the results of actual measurements.

712.8 Determination of possible need for remedial action where criteria have not been met.

The possible need for remedial action may be determined where the criteria in 712.7 have not been met if various other factors are present. Such factors include, but are not necessarily limited to, size of the affected area, distribution of radiation levels in the affected area, amount of tailings, age of individuals occupying affected area, occupancy time, and use of the affected area.

712.9 Factors to be considered in determination of order of priority for remedial action.

In determining the order or priority for execution of remedial action, consideration shall be given, but not necessarily limited to, the following factors:

(a) Classification of structure. Dwellings and schools shall be considered first.

(b) Availability of data. Those structures for which data on indoor radon daughter concentration levels and/or external gamma radiation levels are available when the program starts and which meet the criteria in 712.7 will be considered first.

APPENDIX 6
(cont'd)

(c) Order of application. Insofar as feasible remedial action will be taken in the order in which the application is received.

(d) Magnitude of radiation level. In general, those structures with the highest radiation levels will be given primary consideration.

(e) Geographical location of structures. A group of structures located in the same immediate geographical vicinity may be given priority consideration particularly where they involve similar remedial efforts.

(f) Availability of structures. An attempt will be made to schedule remedial action during those periods when remedial action can be taken with minimum interference.

(g) Climatic conditions. Climatic conditions or other seasonable considerations may affect the scheduling of certain remedial measures.

712.10 Selection of appropriate remedial action.

(a) Tailings will be removed from those structures where the appropriately averaged external gamma radiation level is equal to or greater than 0.05 mR/h above background in the case of dwellings and schools and 0.15 mR/h above background in the case of other structures.

(b) Where the criterion in paragraph (a) of this section is not met, other remedial action techniques, including but not limited to sealants, ventilation, and shielding may be considered in addition to that of tailings removal. ERDA shall select the remedial action technique or combination of techniques, which it determines to be the most appropriate under the circumstances.

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